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Uncertainty and Sensitivity Analysis of Hot Leg LOCA in Two-Loop PWR Using RELAP5 Version 3.3lj

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ABSTRACT

The reactor pressure vessel (RPV) is vulnerable to occurrence of pressurised thermal shock (PTS) in a pressurised water reactor (PWR). PTS can occur during several overcooling scenarios, including loss of coolant accidents. Thermal hydraulic calculations of overcooling scenarios provide an input to structural analysis. The purpose of this study is to perform the calculation of hot leg small break loss of coolant accident with uncertainty and sensitivity analysis. The reactor selected was two-loop PWR, for which verified and validated input deck was available. For analysis the RELAP5 developmental version 3.3lj from 2022 with built-in code uncertainty parameters has been used. For uncertainty and sensitivity analysis Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Data Processing program Software for Uncertainty and Sensitivity Analyses (SUSA) Version 4.2.5 has been used. For uncertainty and sensitivity analysis 208 RELAP5 runs have been performed. The three figures of merit selected for uncertainty analysis were reactor pressure, liquid temperature and reactor vessel wall temperature below the cold leg connection.

The results are presented for reference calculation, sensitivity study varying one parameter at a time and uncertainty and sensitivity analysis.

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EXECUTIVE SUMMARY

The reactor pressure vessel (RPV) is vulnerable to occurrence of pressurised thermal shock (PTS) in a pressurised water reactor (PWR). PTS can occur during several overcooling scenarios, including loss of coolant accidents. Thermal hydraulic calculations of overcooling scenarios provide an input to structural analysis. The purpose of this study is to perform the calculation of hot leg small break loss of coolant accident with uncertainty and sensitivity analysis. The reactor selected was two-loop PWR, for which verified and validated input deck was available. For analysis the RELAP5 developmental version 3.3lj from 2022 with built-in code uncertainty parameters has been used. For uncertainty and sensitivity analysis Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Data Processing program Software for Uncertainty and Sensitivity Analyses (SUSA) Version 4.2.6 has been used. For uncertainty and sensitivity analysis 208 RELAP5 runs have been performed. The reason for this is that three figures of merit selected for uncertainty analysis were selected: reactor pressure, liquid temperature and reactor vessel wall temperature below the cold leg connection. With 208 runs two-sided tolerance limits could be calculated for scalar values with 95 % confidence level and 95 % probability.

The results are presented first for the reference calculation of hot leg loss of coolant accident. In reference calculation best estimate values of input uncertain parameters were used. Before uncertainty analysis the sensitivity study varying one parameter at a time has been performed for 15 selected input uncertain parameters. In the uncertainty analysis minimum and maximum values in each time step were determined, representing lower and upper uncertainty bound, respectively. In the sensitivity analysis the four correlation based sensitivity indices and Sobol's first order index for output uncertain parameters have been used.

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ABBREVIATIONS AND ACRONYMS

AFW	auxiliary feedwater
ACC	accumulator
APAL	Advanced PTS Analysis for LTO
CL	cold leg
DEC	design extension conditions
FOM	figure of merit
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
HF	Henry-Fauske
HL	hot leg
HPSI	high pressure safety injection
HTC	heat transfer coefficient
IAEA	International Atomic Energy Agency
LOCA	loss of coolant accident
LPSI	low pressure safety injection
LTO	long term operation
MD	motor driven
MFW	main feedwater
NPP	nuclear power plant
PRZ	pressurizer
PTS	pressurised thermal shock
PWR	pressurized water reactor
RCP	reactor coolant pump
RHWG	Reactor Harmonisation Working Group
RPV	reactor pressure vessel
RS	respone surface
SG	steam generator
SI	safety injection
SNAP	Symbolic Nuclear Analysis Package
SUSA	Software for Uncertainty and Sensitivity Analyses
U.S. NRC	U.S. Nuclear Regulatory Commission

1 INTRODUCTION

The reactor pressure vessel (RPV) is vulnerable to occurrence of pressurised thermal shock (PTS) in a pressurised water reactor (PWR). PTS can occur during several overcooling scenarios, including loss of coolant accidents. Thermal hydraulic calculations of overcooling scenarios provide input to structural analysis. In this study hot leg small break loss of coolant accident has been simulated with uncertainty and sensitivity analysis. The reactor selected was two-loop PWR, for which verified and validated input deck was available. For analysis the RELAP5 developmental version 3.3lj from 2022 with built-in code uncertainty parameters has been used. For uncertainty and sensitivity analysis Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Data Processing program Software for Uncertainty and Sensitivity Analyses (SUSA) Version 4.2.6 has been used. For uncertainty and sensitivity analysis 208 RELAP5 runs have been performed. The reason for this is that three figures of merit for uncertainty analysis were selected: reactor pressure, liquid temperature and reactor vessel wall temperature below the cold leg connection. With 208 runs two-sided tolerance limits could be calculated for scalar values with 95 % confidence level and 95 % probability.

The report is organized as follows. In Section 2 the RELAP5 input model and scenarios are first described. Then, input uncertain parameters with distributions, and uncertainty and sensitivity analysis method are described. Section 3 first present the results for the reference calculation of hot leg loss of coolant accident. Before the uncertainty analysis a sensitivity study varying one parameter at a time has been performed for 15 selected input uncertain parameters. In the uncertainty analysis using results of 208 runs, minimum and maximum values in each time step were determined, representing lower and upper uncertainty bound, respectively. In the sensitivity analysis the calculated four correlation based sensitivity indices and Sobol's first order index for output uncertain parameters are presented. Finally, the conclusions are given in Section 4.

2 RELAP5 INPUT MODEL, SCENARIOS AND METHODS DESCRIPTION

First the RELAP5 input model is briefly described for a two-loop PWR, followed by simulated scenario description. Then input uncertain parameters are described. Finally, the uncertainty and sensitivity analysis methods used are described.

2.1 <u>RELAP5 Input Model</u>

For calculations RELAP5/MOD3.3lj code version [1] has been used applying the RELAP5 input model of two-loop PWR that has already been used in other studies [2], [3]. A two loop PWR reactor power is 1994 MW and the nuclear steam supply system (NSSS) power is 2000 MW. The base model shown in Figure 2-1 consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points.

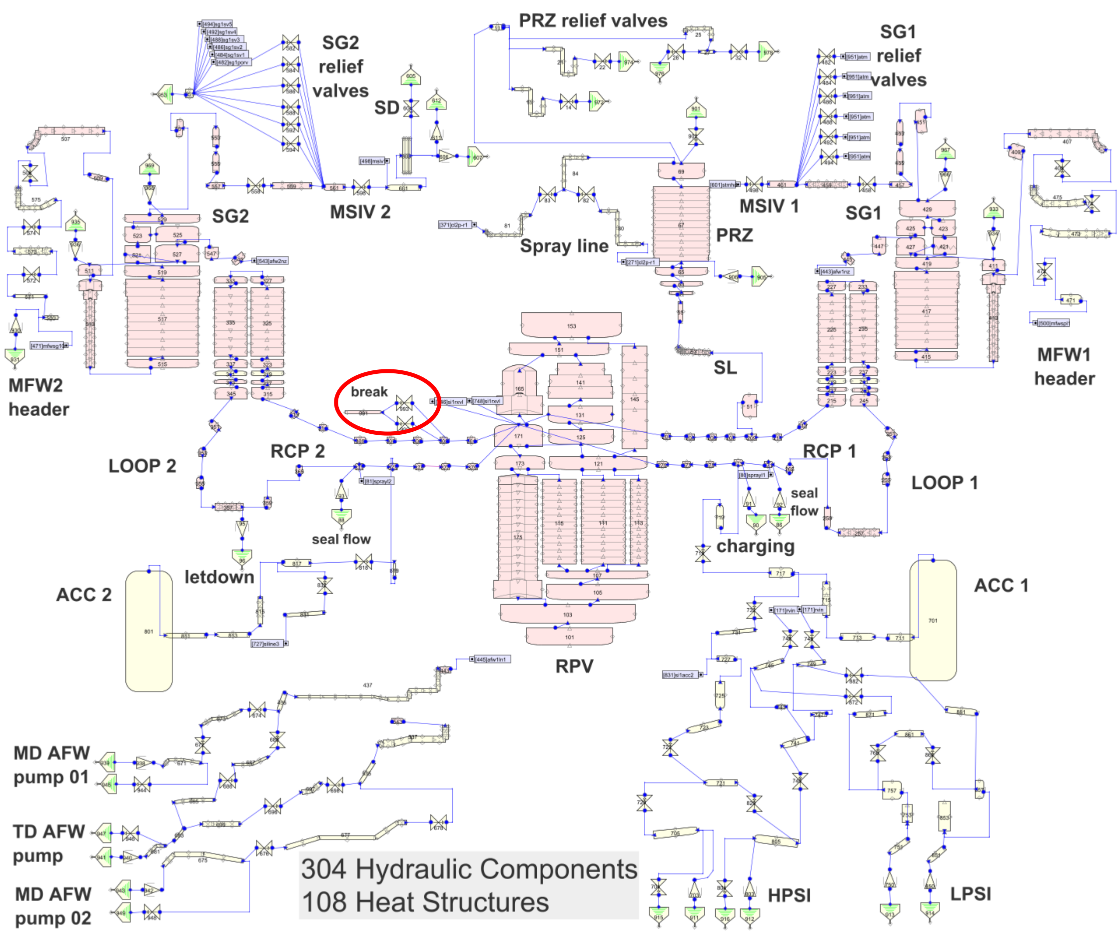

Figure 2-1 RELAP5 Two-Loop PWR Hydraulic Components View

In terms of SNAP (Symbolic Nuclear Analysis Package) this gives 304 hydraulic components and 108 heat structures. Hydraulic components in SNAP consist of both volumes and junctions, where pipe with more volumes is counted as one component. Each heat structure in SNAP connected to pipe is also counted as one component and not as many heat structures as pipe has volumes like it is counted in the RELAP5 output file. This explains the difference in the number of heat structures in Figure 2-1 and that reported in the RELAP5 output file. Besides, control variables and logical conditions (trips) represent the instrumentation, regulation (rod control, pressurizer (PRZ) level and pressure, steam dump, steam generator (SG) pressure, etc.) and protection systems (reactor protection, main feedwater (MFW) isolation, safety injection (SI) and auxiliary feedwater (AFW) triggering logic, steam line isolation, etc.). The break is located in the hot leg in the loop without pressurizer.

2.2 Scenarios Description

Initiating event is 45.6 cm^2 (7.62 cm - 3" diameter) break loss of coolant accident (SB LOCA) in the hot leg (HL), which occurred at 0.01 s at full reactor power. Initial and boundary conditions are shown in Table 2-1. A loss of alternate current has been assumed to occur at the same time as the break occurrence. Therefore, the reactor coolant pumps (RCPs) trip immediately. One train of active emergency core cooling systems is available (one high pressure injection pump, one low pressure injection pump) and both accumulators. Both motor driven (MD) auxiliary feedwater (AFW) pumps were assumed available.

After break occurrence the reactor trips on (compensated) low pressurizer pressure signal (12.99 MPa), which further causes the turbine trip. The safety injection (SI) signal is generated on the low-low pressurizer pressure signal at 12.27 MPa. On SI signal active safety systems like high pressure safety injection (HPSI) pump, low pressure safety injection (LPSI) pump and both MD AFW pumps start. HPSI pumps start to inject with 5 s delay on SI signal. When primary pressure drops below 4.96 MPa, both accumulators start to inject. When primary pressure drops below 1.13 MPa, LPSI pumps start to inject.

Table 2-1 Initial and Boundary Conditions

Parameter	Two-loop PWR	RELAP5
Core power (MW)	1994	1994
Steam generator power (MW)	1000	996.5 / 1002.5
Pressurizer pressure (MPa)	15.51	15.51
Steam generator pressure (MPa)	6.28	6.44 / 6.44
Cold leg temperature (K)	559.2	559.51 / 559.32
Hot leg temperature (K)	596.9	596.79 / 596.79
Feedwater temperature (K)	492.6	492.5
Pressurizer level (%)	55.7	55.8
Steam generator narrow range level (%)	69.3	69.3 / 69.3
Steam mass flow (kg/s)	544.5	541.3 / 544.5

2.3 Input Uncertain Parameters with Distributions

In total 15 input uncertain parameters have been selected based on the results obtained in the frame of the APAL project [4]. The following four input uncertain parameters used in APAL uncertainty study have been considered not much significant in our study for selected HL LOCA: secondary side pressure, ACC initial nitrogen volume, and HPSI and LPSI pump pressure curve multipliers. The reference case consists of values shown in column four (reference or best estimate values).

Table 2-2 Input Uncertain Parameters with Distributions

Par. No.	Parameter Name	Unit	Ref. Value / Best Estimate	Distribution Type	Distribution Parameter1	Distribution Parameter2	Minimum	Maximum
1	Core power	W	1994.0E6	Normal	1994.0E6	19.94E6	-infinity (1974.1E6)[1]	infinity (2013.9E6)[1]
2	Pressurizer pressure	Pa	15.512E6	Normal	15.512E6	0.15512E6	-infinity (15.357E6)[1]	infinity (156.671E6)[1]
3	Decay heat	-	1	Uniform	0.9	1.1	0.9	1.1
4	Timing of SIS actuation	s	5	Uniform	0	20	0	20
5	ACC injection temperature	K	322	Uniform	312	332	312	332
6	ACC initial pressure	Pa	4.928E+06	Uniform	4.728E+06	5.128E+06	4.728E+06	5.128E+06
7	HPSI temperature	K	310	Uniform	295	325	295	325
8	HP pump flow curve	-	1	Normal	1.0	0.1	-infinity (0.9)[1]	infinity (1.1)[1]
9	LP pump flow curve	-	1	Normal	1.0	0.1	-infinity (0.9)[1]	infinity (1.1)[1]
10	Initial pressurizer level	%	55.7	Uniform	48.34	63.06	48.34	63.06
11	Thermal-nonequilibrium coefficient for Henry-Fauske model	-	0.93	Weibull	7	1	0	1.5
12	Single-phase liquid to wall HTC	-	1	Log. Uniform	0.8	1.2	0.8	1.2
13	Single-phase vapour to wall HTC	-	1	Log. Uniform	0.8	1.2	0.8	1.2
14	Wall-drag coefficient	-	1	Log. Uniform	0.5	2	0.5	2
15	Form-loss coefficient	-	1	Log. Uniform	0.9	1.1	0.9	1.1

2.4 Uncertainty and Sensitivity Analysis Method

In the uncertainty analysis the input uncertain parameters are propagated through the computer code model [5]. Multiple simulations of the transient scenario produce multiple sets of simulation output, each set of output is the result of a unique combination of randomly-chosen values for the input parameters. Different techniques for the uncertainty propagation in the thermalhydraulic code calculations were identified in the world, including Monte Carlo analysis, response surface (RS) methods and statistical tolerance limits [6]. Due to demanding calculation requirements the Monte Carlo method is currently not applicable to the complex thermalhydraulic codes. In the RS methods the response surface replaces the code calculation in the Monte Carlo analysis. The statistical tolerance limits is approach with a random sampling input parameters N times and then use the computer code directly to generate N outputs which are used to estimate the actual uncertainty.

[1] Value used in the sensitivity study

Statistical tolerance limits are defined as the limits of an interval for which it can be stated with a given level of confidence that the interval contains at least a specified proportion of the population. These limits are found based on information from a sample. The limits are influenced by the sample size, the sample mean, the sample standard deviation and the specified proportion to be within the statistical tolerance limits. Importantly, as the sample size becomes large, the statistical tolerance limits approach the values found by using the population parameters. In addition, the limits are also influenced by the distribution of the characteristic. In the case of a characteristic with a distribution that is normal, parametric statistical tolerance limits are appropriate. Statistical upper and lower of the distribution of the key (output) parameter are determined as the tolerance limits with a specified probability. When a characteristic is not normal, nonparametric statistical tolerance limits are preferred because they do not depend on the assumption that the data follow a particular distribution.

Nonparametric statistics are not based on a specific distribution. They are often referred to as "distribution free". When the distribution hypothesis is rejected by goodness-of-fit test (it is unknown) it is possible to determine tolerance limits by randomly sampling the character in question. The consideration of nonparametric tolerance limits was originally presented by Wilks. Wilks study showed that for continuous populations, the distribution of P, the proportion of the population between two order statistics from a random sample, is independent of the population sampled.

In this case the tolerance limit is given by order statistics (highest, second highest etc.). When the tolerance limit is given by the maximum order statistics, the required minimum number of calculations (N) is given by Wilks formula:

$$1 - \gamma^N - N(1-\gamma)\gamma \geq \beta \tag{1}$$

for two-sided tolerance limit (specifying $\gamma = \beta$=0.95 minimum N becomes 93) and

$$1 - \gamma^N \geq \beta \tag{2}$$

for one-sided tolerance limit (specifying $\gamma = \beta$=0.95 minimum N becomes 59), where γ is desired probability content and β is confidence level.

GRS was the first one in thermalhydraulic safety analysis, who based on the works of Wilks established methodology called GRS. GRS methodology could be used also for more figures of merits (FOMs), when they are independent.

The confidence level is specified because the probability content is not analytically determined. It accounts for the possible influence of the sampling error due to the fact that the statements are obtained from a random sample of limited size. Thus, the number N of calculation runs depends only on the desired probability content γ and confidence level β of the statistical tolerance limits.

The advantage of nonparametric statistics is that the number of input uncertain parameters is limited only by the ability of the user to define and implement uncertainties and that all uncertainties are propagated together. The weakness is that it does not produce probability distributions and that tolerance limits are conservatively bounding the real limits.

When the number of dependent FOMs is greater than one, the following Wilks formula obtained by Wald (briefly described by E. Zugazagoitia et al. in [7]) can be used:

$$\beta = \sum_{j=0}^{N-R^*} \binom{N}{j} \gamma^j \cdot (1-\gamma)^{N-j} \tag{3}$$

$$R^* = p \cdot \sum_{i=1}^{R} d_i \tag{4}$$

Where R is the number of FOMs, d_i is the number of bounds of the FOMs (upper and/or lower), and p is order.

Examples of the minimum number of samples for one- and two-sided tolerance limit, when probability γ is 0.95 and confidence level β is at least 0.95 (in general it is slightly higher to satisfy criterion) are shown in Table 2-3.

Table 2-3 Examples of the Minimum Number of Samples for One- and Two-Sided Tolerance Limit (γ=0.95, β=0.95)

R	N (γ=0.95)	confidence level β, one-sided	N (γ=0.95)	confidence level β, two-sided
1	59	0.951506	93	0.950024
2	93	0.950024	153	0.950555
3	124	0.950470	208	0.950774
4	153	0.950555	260	0.950192
5	181	0.950837	311	0.950345
6	208	0.950775	361	0.950566
7	234	0.950144	410	0.950547
8	260	0.950192	458	0.950154
9	286	0.950715	506	0.950307
10	311	0.950350	554	0.950850

In this study three FOMs were used in the uncertainty analysis: (a) primary pressure, (b) liquid temperature below reactor vessel CL inlet, and (c) reactor vessel wall temperature below CL inlet. For three FOMs the one-sided approach requires at least 124 samples, while for two-sided approach 208 samples are needed. On the other hand, 208 samples give also sufficient samples to consider 6 FOMs with one-sided approach. As this analysis is to be input to fracture mechanics analysis (like example in [8]), an input would be 208 calculated trends of three output uncertain parameters calculated by RELAP5 rather than upper and lower uncertainty bounds calculated by uncertainty methodology. Namely, in the APAL project it has been already shown that selecting maxima for pressure and minima for temperatures are not useful for fracture mechanics uncertainty quantification. Therefore, the emphasis in this study was to provide 208 runs using random sampled input uncertain parameters and sensitivity analysis of output uncertain parameters based on 208 runs.

According to [5] a sensitivity analysis or, more precisely, an uncertainty importance analysis helps to identify those uncertain input parameters which mainly contribute to the uncertainty of the computational result. SUSA software implemented four correlation based sensitivity indices applicable to individual parameters. These are:

− Pearson's correlation,

− Spearman's rank correlation,

− Blomqvist’s medial correlation,

− Kendall's rank correlation.

The classical methods are described in [5] the following:

“Pearson’s correlation coefficient (or Ordinary correlation coefficient) characterizes the bivariate Normal distribution and, therefore, is often employed to model dependency between two parameters *X* and *Y*.” For more detail refer to [5].

“Blomqvist’s medial correlation coefficient (or Blomqvist’s beta or population quadrant measure) is a practical approach to take into account a degree of association between parameters without structural information about the distribution of the corresponding parameters. This measure enables the analyst to practically take into account the experts’ belief about the effect of an increasing parameter *X* on a parameter *Y* relative to the medians m_X and m_Y of the assigned distributions.” For more details refer to [5].

“Kendall’s rank correlation coefficient (or Kendall’s tau) is a practical approach to take into account a degree of association between two parameters *X* and *Y* and may be considered as an extension of Blomqvist’s medial correlation coefficient. Both measures of association do have the same properties, only the choice of the reference point for the concordance deliberately differs. Instead of the fixed reference point in terms of the pair of medians (m_X, m_Y) for Blomqvist’s measure, another bivariate parameter pair (X_2, Y_2) is employed for Kendall’s measure.” For more details refer to [5].

“Spearman’s rank correlation coefficient (or Spearman’s rho) is a practical approach to take into account a degree of association between two random parameters (*X*, *Y*) and may be considered as an extension of Kendall’s rank correlation coefficient.” For more details refer to [5].

Standardized regression coefficients of above four ordinary sensitivity indices and rank of standardized regression coefficient of above four ordinary sensitivity indices are not shown in this study (provided by SUSA), similarly partial and rank of partial rank coefficient of above four ordinary sensitivity indices (also provided by SUSA).

Besides correlation related sensitivity measures, the classical correlation ratio from original and rank transformed data may serve as sensitivity index. The square of the correlation ratio is equivalent to the variance based first order sensitivity index also known as Sobol’s first order index. These Sobol’s first order indices are shown in this report.

The Sobol Index is described in [5] the following:

“A well-known approach to calculate sensitivity indices offers the framework of the Sobol indices (SI) as variance-based sensitivity measures. The SIs describe the sensitivity patterns of a model via the full decomposition of the variance of the model response into terms depending on the model input parameters and their interactions. A high S_j (Sobol’s first order sensitivity index or uncertainty importance measure with respect to parameter X_j) value indicates that parameter X_j strongly influences the variance $Var(Y)$.” For more details refer to [5].

3 RESULTS

Sections 3.1 through 3.4 show the results of reference case scenario calculation, sensitivity study, uncertainty analysis and sensitivity analysis, respectively.

3.1 Reference Case Scenario

The sequence of events for reference case is shown in Table 3-1. The results of reference case scenario simulations are shown in Figures 3-1 and 3-2.

Table 3-1 Sequence of Events for Reference Case

Event	Time (s)
Break occurrence	0.01
Reactor trip signal	2.37
Turbine trip	2.37
Safety injection signal	18.89
Main feedwater pump trip	18.9
High pressure safety injection	23.89
Auxiliary feedwater start	23.9
Accumulator injection	905
Low pressure injection	2660

The reference case value for thermal-hydraulic non-equilibrium constant in Henry-Fauske (HF) choke flow model was not selected to be default value 0.14, because it is very low probability to be sampled this value of parameter having Weibull distribution. Therefore value 0.93 has been selected, which represents approximately 50 percentile of Weibull distribution. However, it should be noted that this change of value has quite large influence on the results shown in Figures 3-1 and 3-2. The break flow is initially higher (see Figure 3-2(a)), therefore the pressure (Figure 3-1(a)) and temperatures drop is faster (see Figure 3-1(b) and Figure 3-1(c), respectively). Due to faster pressure drop, there is earlier injection of HPSI pump, accumulator and LPSI pump shown in Figures 3-2(d), 3-2(e) and 3-2(f), respectively. Finally, break flow and injection flows have influence on primary mass inventory, shown in Figure 3-2(b).

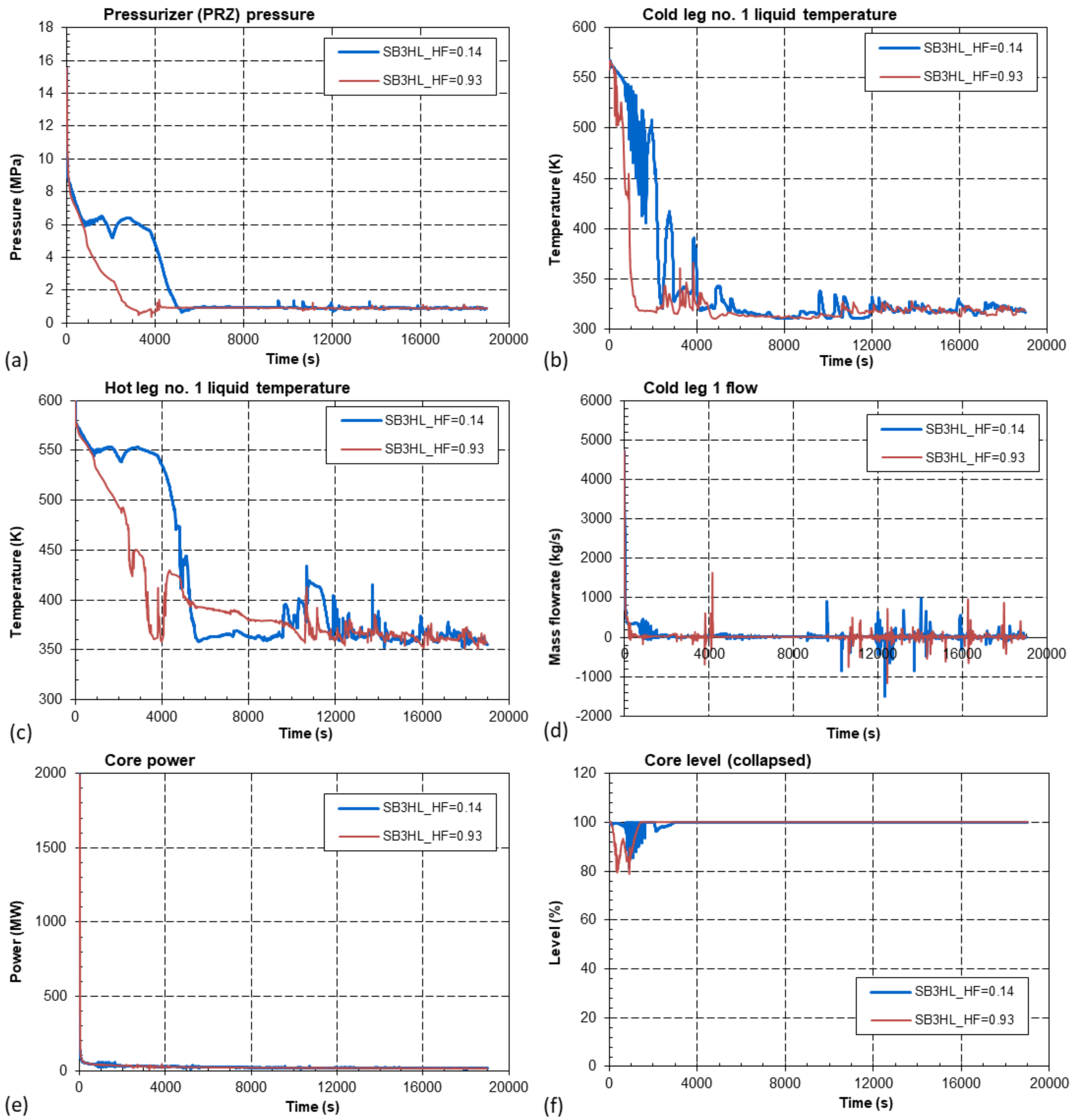

Figure 3-1 RELAP5 Results for Reference Case and Case with Default HF Choke Flow Values (Part 1)

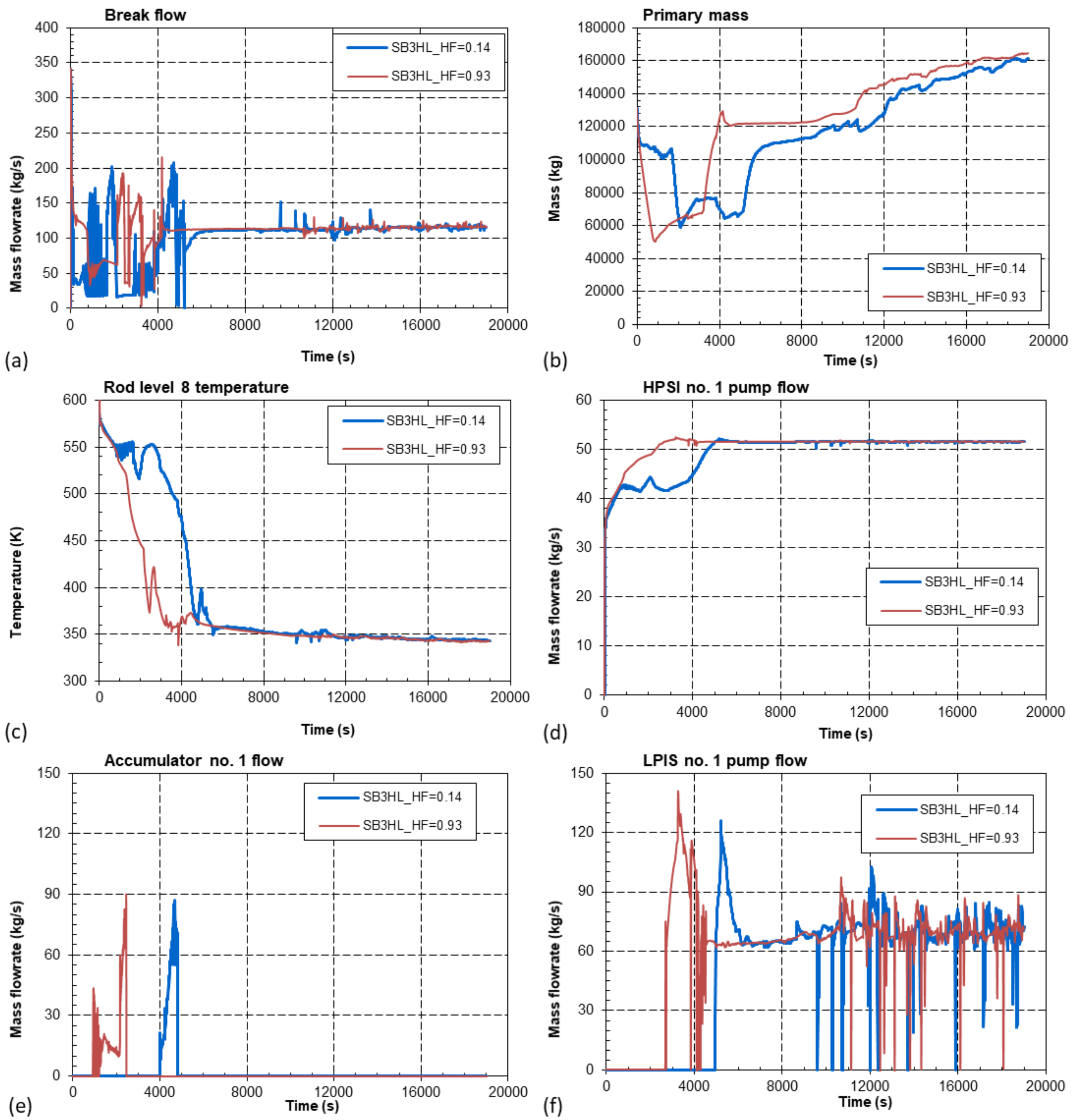

Figure 3-2 RELAP5 Results for Reference Case and Case with Default HF Choke Flow Values (Part 2)

3.2 Sensitivity Study

Before uncertainty analysis has been performed, sensitivity study varying one parameter at a time has been performed following typical U.S. NRC approach for uncertainty quantification [9]. It should be noted that sensitivity study varying one parameter at a time is not sensitivity analysis. In the sensitivity study minimum and maximum values of uncertain input parameters shown in Table 2-2 were used. In this way the reader can get preliminary information on the impact of selected 15 uncertain input parameters on the FOMs (note that compensating effects of different parameters are not taken into account in the sensitivity study varying one parameter at a time). The results are shown in Figures 3-3 through 3-8.

From Figures 3-3 and 3-4 it can be seen that for primary pressure the most sensitive input uncertain parameters are ‘Par2’ (initial pressurizer pressure) and ‘Par11’ (thermal-nonequilibrium coefficient for Henry-Fauske choke flow model). In the time interval 0-5000 s sensitive input uncertain parameters are also ‘Par3’ (decay heat) and ‘Par15’ (form-loss coefficient) as shown in Figures 3-3 and 3-4.

From Figures 3-5 and 3-6 it can be seen that for liquid temperature below RPV cold leg inlet the most sensitive input uncertain parameters are ‘Par2’ (initial pressurizer pressure), 'Par 7' (HPSI temperature) and ‘Par11’ (thermal-nonequilibrium coefficient for Henry-Fauske choke flow model). In the time interval 0-5000 s sensitive input uncertain parameters are also ‘Par3’ (decay heat) and ‘Par15’ (form-loss coefficient) as shown in Figures 3-5 and 3-6.

From Figures 3-7 and 3-8 it can be seen that for wall temperature below RPV cold leg inlet the most sensitive input uncertain parameters are ‘Par2’ (initial pressurizer pressure), 'Par 7' (HPSI temperature) and ‘Par11’ (thermal-nonequilibrium coefficient for Henry-Fauske choke flow model). Sensitive input uncertain parameters are also ‘Par3’ (decay heat) and ‘Par15’ (form-loss coefficient) as shown in Figures Figures 3-7 and 3-8.

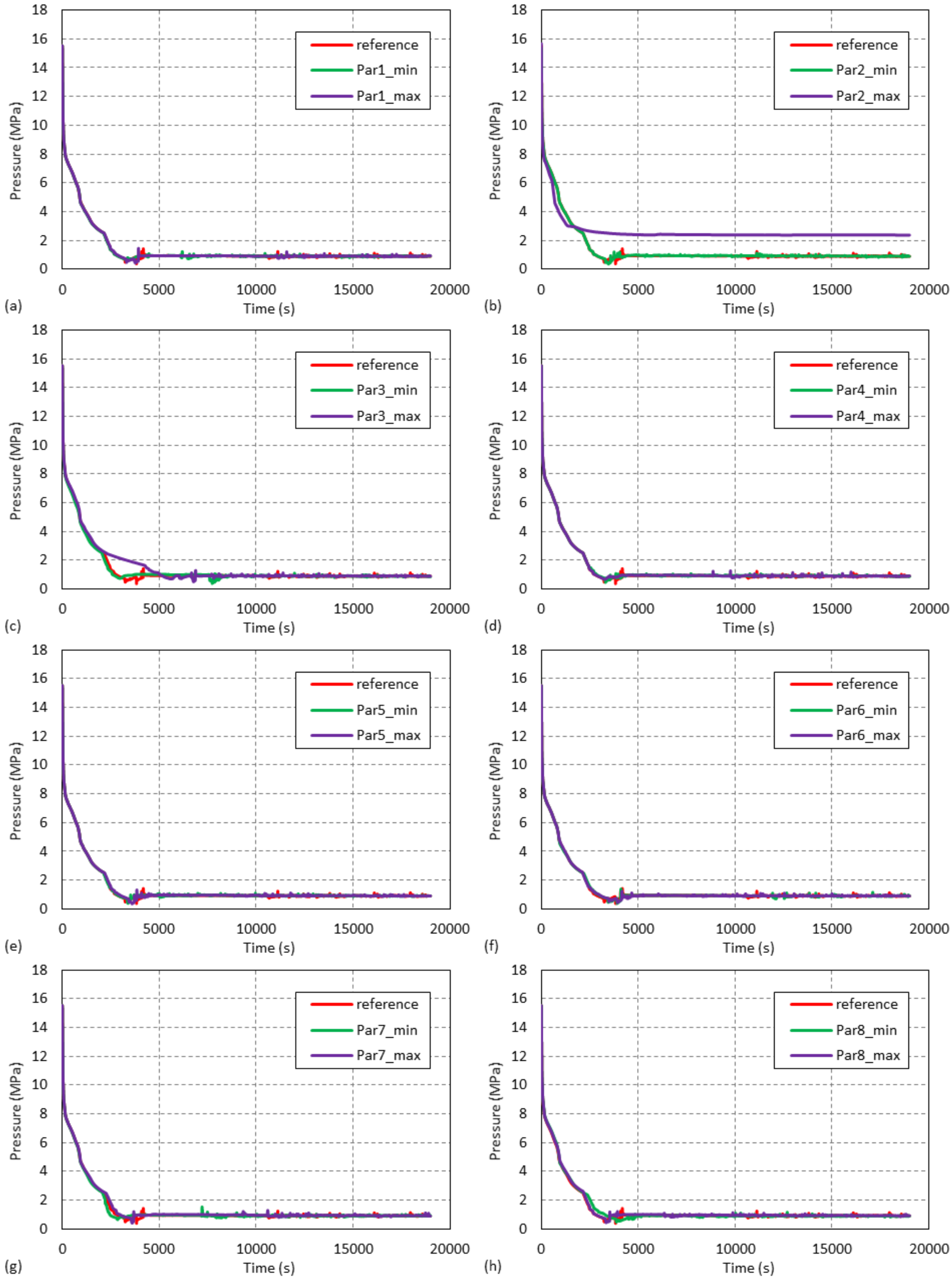

Figure 3-3 Impact of Uncertain Input Parameters 1 through 8 on Primary Pressure

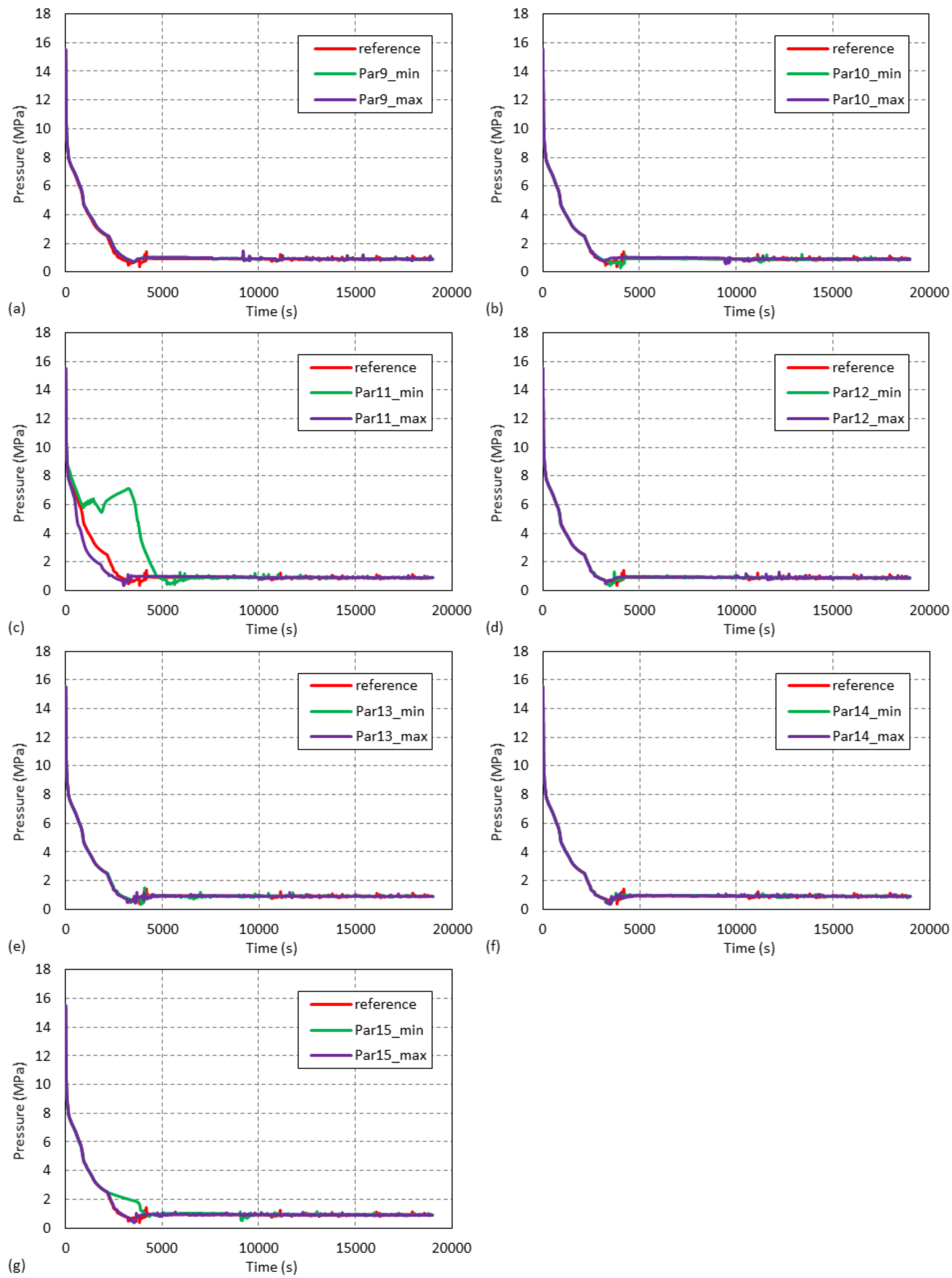

Figure 3-4 Impact of Uncertain Input Parameters 9 through 15 on Primary Pressure

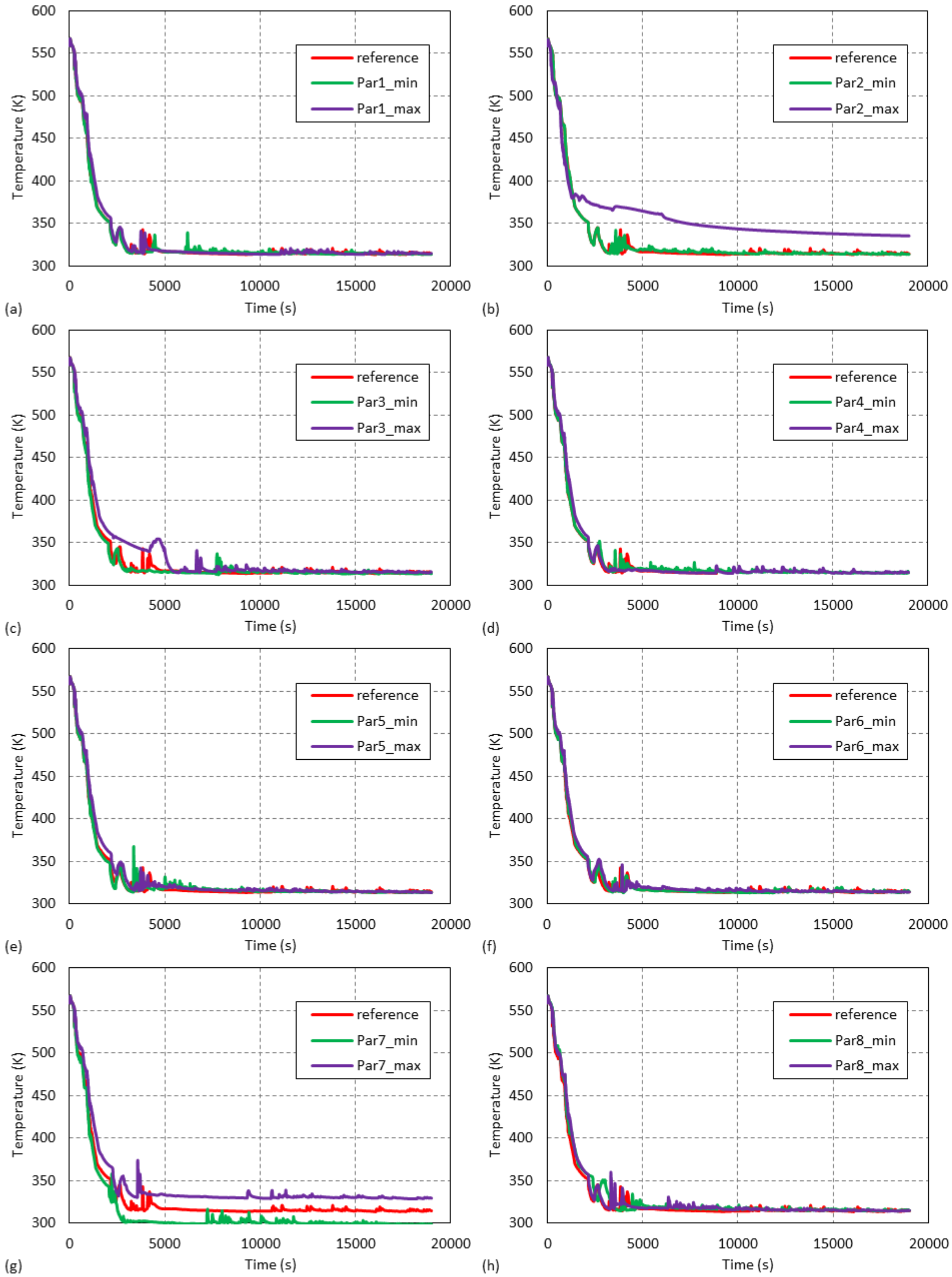

Figure 3-5 Impact of Uncertain Input Parameters 1 through 8 on Liquid Temperature below RPV Cold Leg Inlet

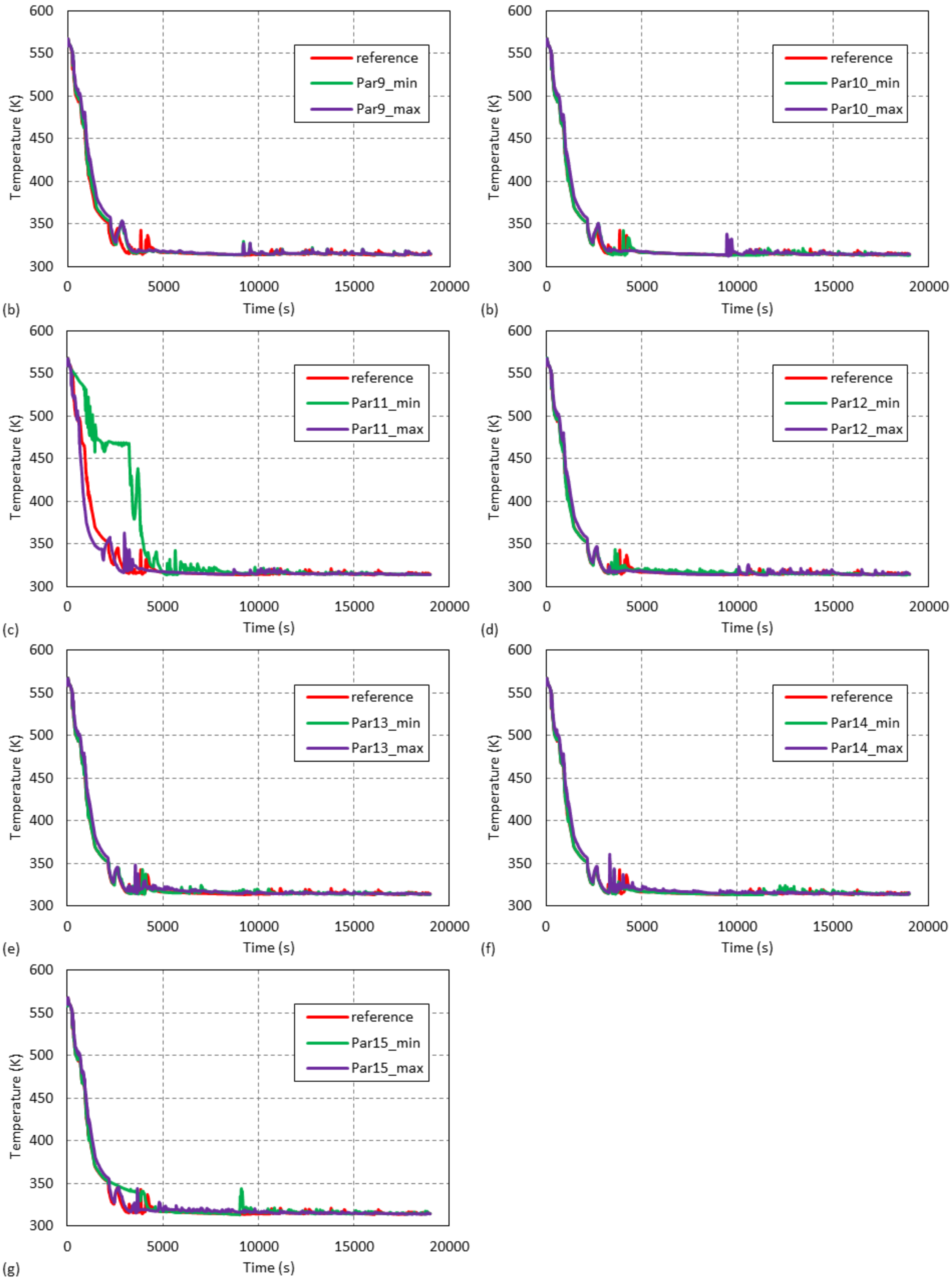

Figure 3-6 Impact of Uncertain Input Parameters 9 through 15 on Liquid Temperature below RPV Cold Leg Inlet

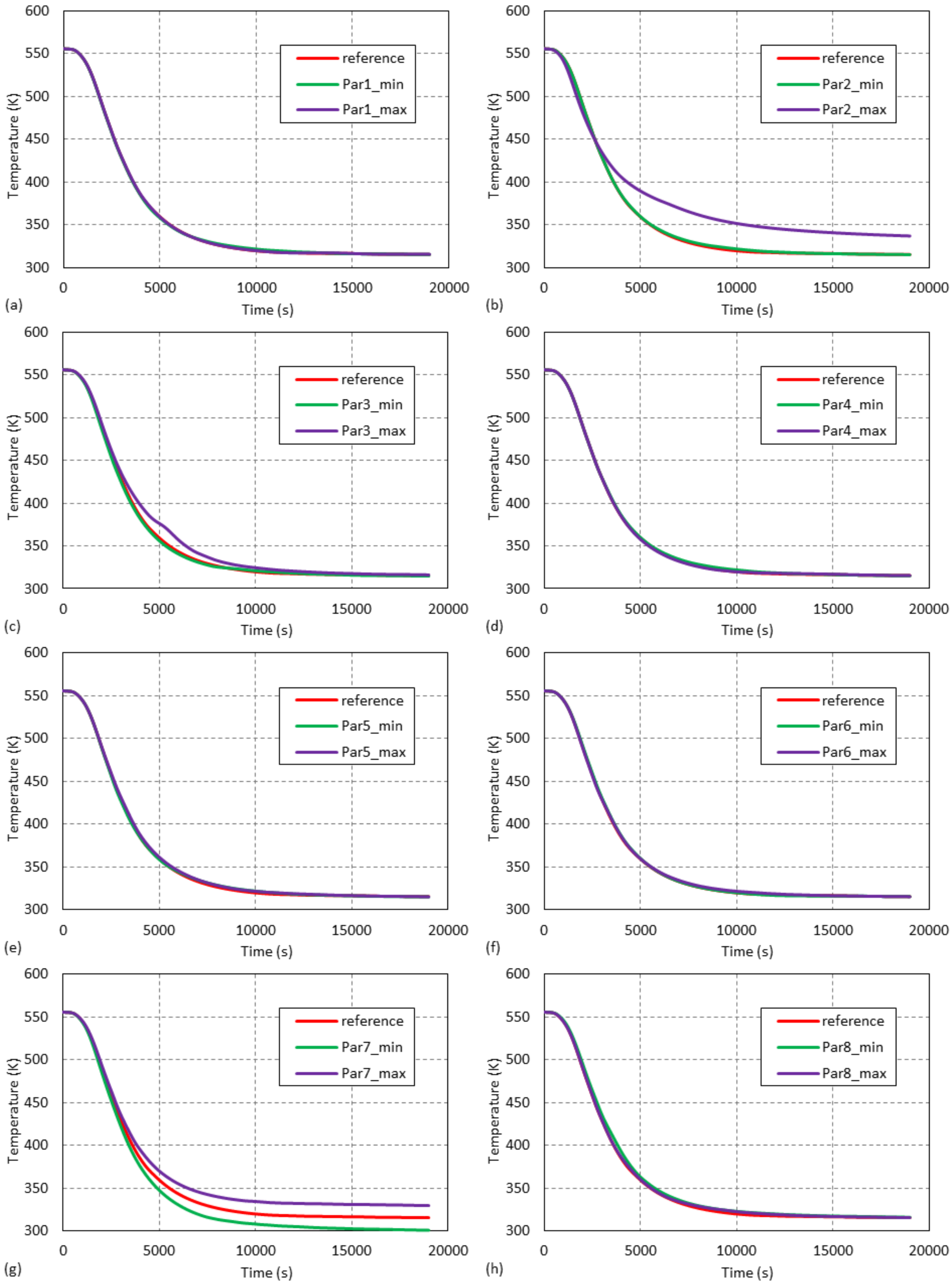

Figure 3-7 Impact of Uncertain Input Parameters 1 through 8 on Wall Temperature below RPV Cold Leg Inlet

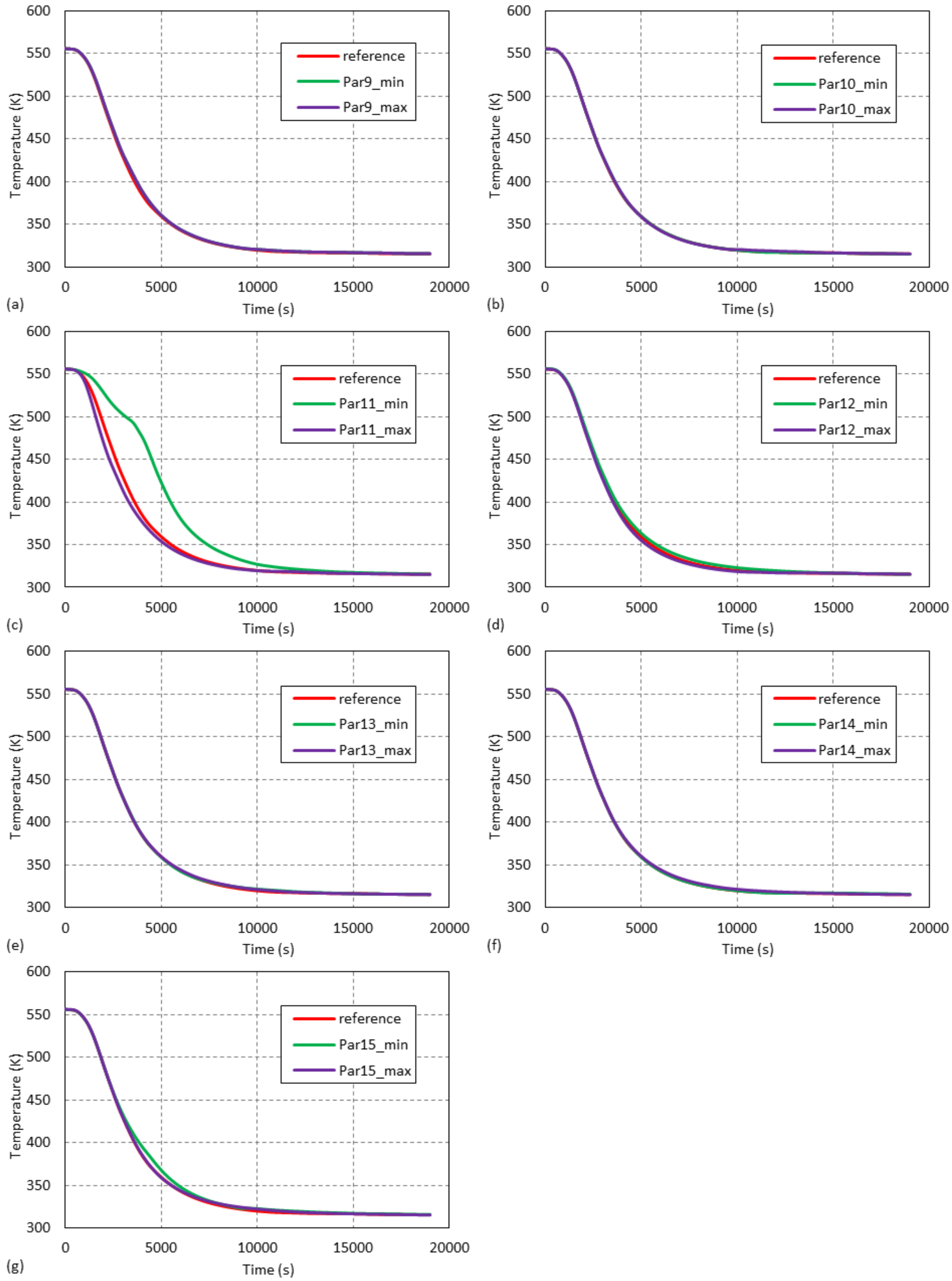

Figure 3-8 Impact of Uncertain Input Parameters 9 through 15 on Wall Temperature below RPV Cold Leg Inlet

3.3 <u>Uncertainty Analysis</u>

Figures 3-9, 3-10, and 3-11 show the uncertainty of a time-dependent results for the FOM1, FOM2, and FOM3, respectively, represented by the time histories obtained from 208 Monte Carlo simulation runs. Figures 3-12, 3-13 and 3-14 show maxima, minima, medians, means and reference curves obtained from 208 runs. It can be seen that in all three figures (i.e. Figures 3-12, 3-13 and 3-14) the reference calculation is bounded by minimum and maximum curve, while there is difference between reference calculation and mean curve.

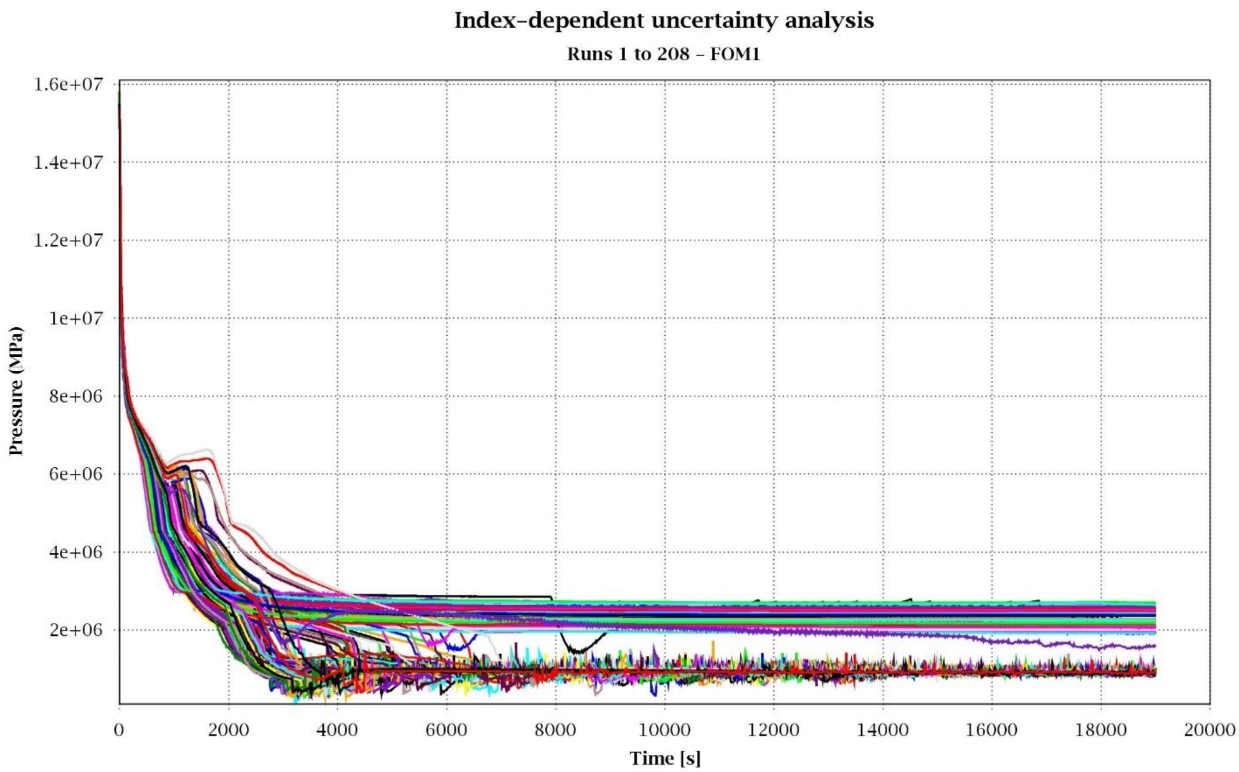

Figure 3-9 Primary Pressure (208 Samples)

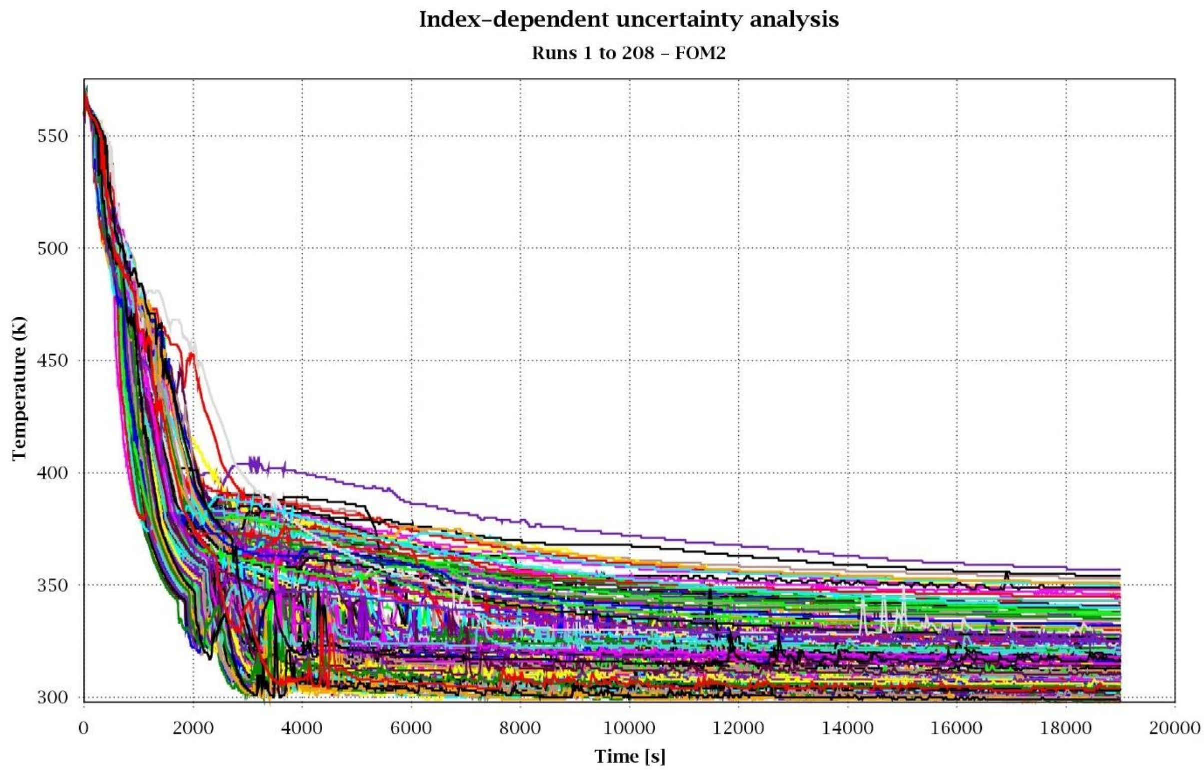

Figure 3-10 Liquid Temperature below RPV Cold Leg Inlet (208 Samples)

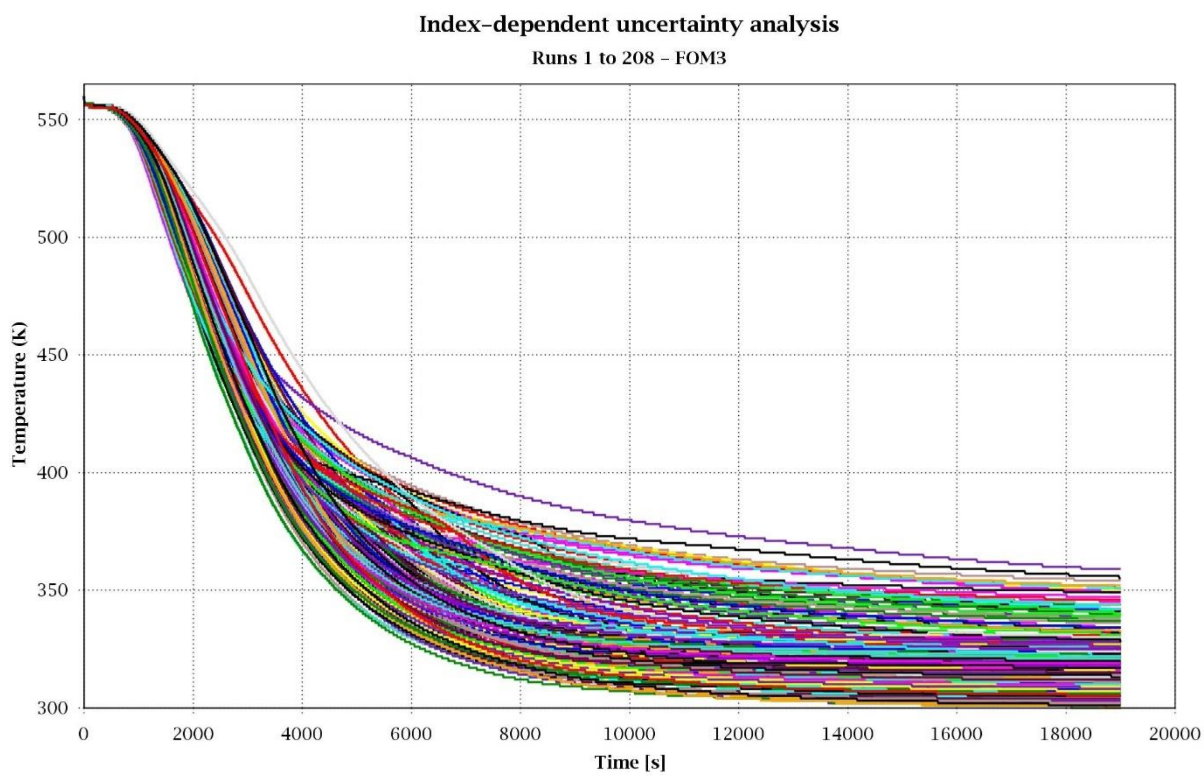

Figure 3-11 Wall Temperature below RPV Cold Leg Inlet (208 Samples)

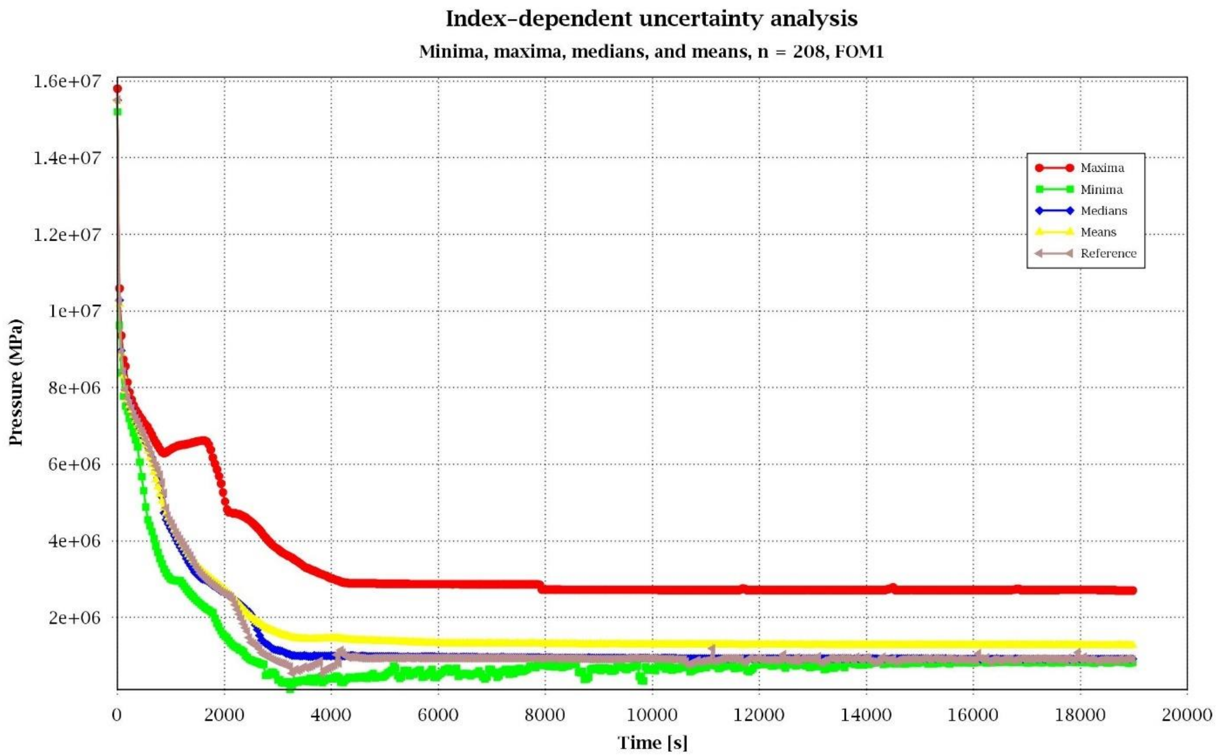

Figure 3-12 Primary Pressure (Maxima, Minima, Medians, Means, Reference)

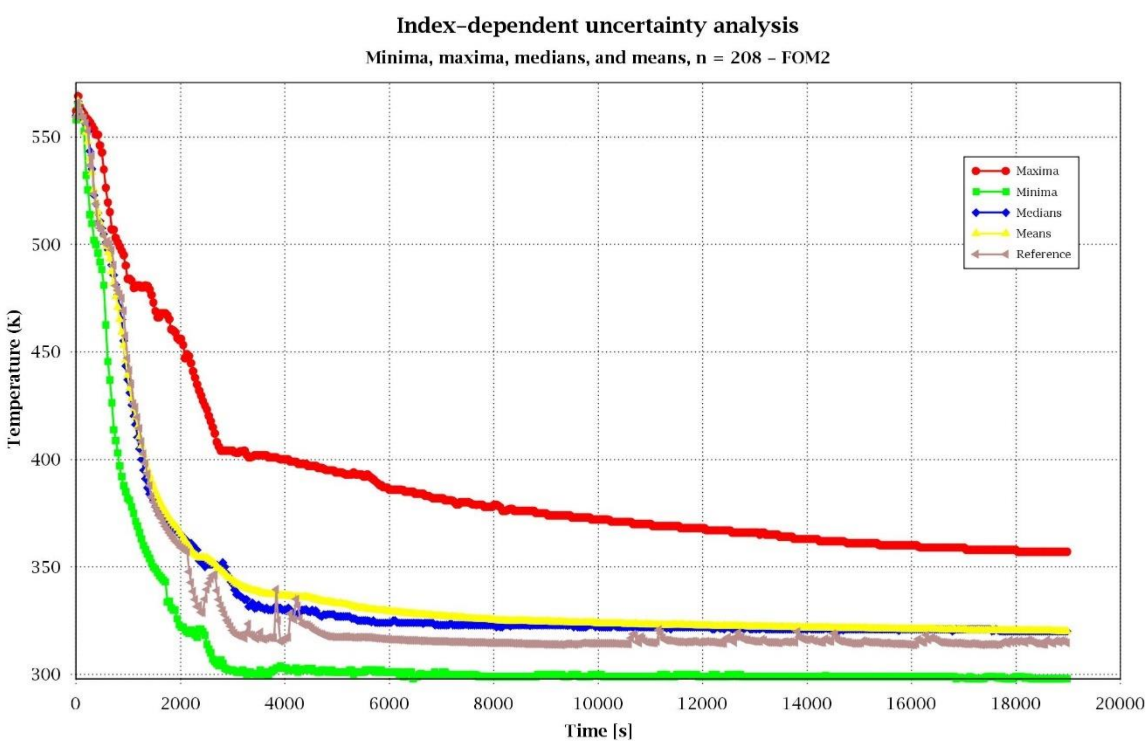

Figure 3-13 Liquid Temperature below RPV Cold Leg Inlet (Maxima, Minima, Medians, Means, Reference)

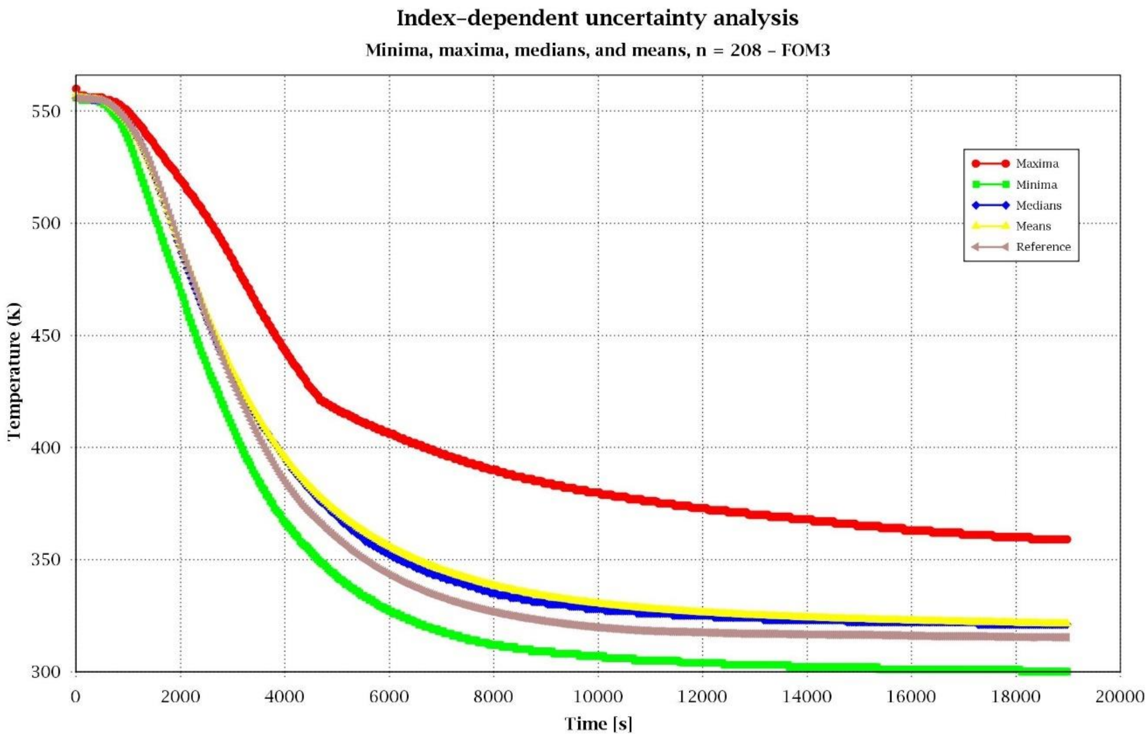

Figure 3-14 Wall Temperature below RPV Cold Leg Inlet (Maxima, Minima, Medians, Means, Reference)

3.4 Sensitivity Analysis

In case of Person's correlation index the dependency between parameters 'Par 2' and 'Par 11', respectively and FOM1 is the largest (see Figure 3-15). Dependency can be also seen 'Par3' and FOM1. This agrees well with the results of sensitivity study (see Figures 3-3 and 3-4). Timing for parameters also agree well. 'Par 2' changes the sign and has all the time significant dependency with FOM1, while 'Par 11' after 5000 s has no more significant dependency with FOM1.

In case of Person's correlation index the dependency between parameters 'Par 2', 'Par 7' and 'Par 11', respectively, and FOM2 and FOM3, respectively, is the largest (see Figures 3-16 and 3-17, respectively). Dependency can be also seen for 'Par3'. This agrees well with the results of sensitivity study (see Figures 3-5 and 3-6 for FOM2, and Figures 3-7 and 3-8 for FOM3). Timing for parameters also agree well. 'Par 2' changes the sign and has all the time significant dependency with FOM2 and FOM3, 'Par7' has all the time positive significant dependency, while 'Par 11' after 5000 s has no more significant dependency with FOM2 and FOM3.

In case of Spearman's rank correlation, Blomqvist's medial correlation and Kendall's rank correlation indices a degree of association between parameters 'Par 2' and 'Par 11' and FOM1 is the largest (see Figure 3-15). This agrees with the results of sensitivity study (see Figures 3-3 and 3-4). Timing for parameters also agree well. 'Par 2' changes the sign and has all the time significant degree of association with FOM1, while 'Par 11' after 5000 s has no more significant a degree of association with FOM1.

In case of Spearman's rank correlation, Blomqvist's medial correlation and Kendall's rank correlation indices a degree of association between 'Par 2', 'Par 7' and 'Par 11' and FOM2 and FOM3 is the largest (see Figure 3-16 and Figure 3-17, respectively). This also agrees well with the results of sensitivity study (see Figures 3-5 and 3-6 for FOM2, and Figures 3-7 and 3-8 for FOM3). Timing for parameters also agree well. 'Par 2' changes the sign and has all the time significant degree of association with FOM2 and FOM3, 'Par7' has all the time positive significant degree of association, while 'Par 11' after 5000 s has no more degree of association with FOM2 and FOM3.

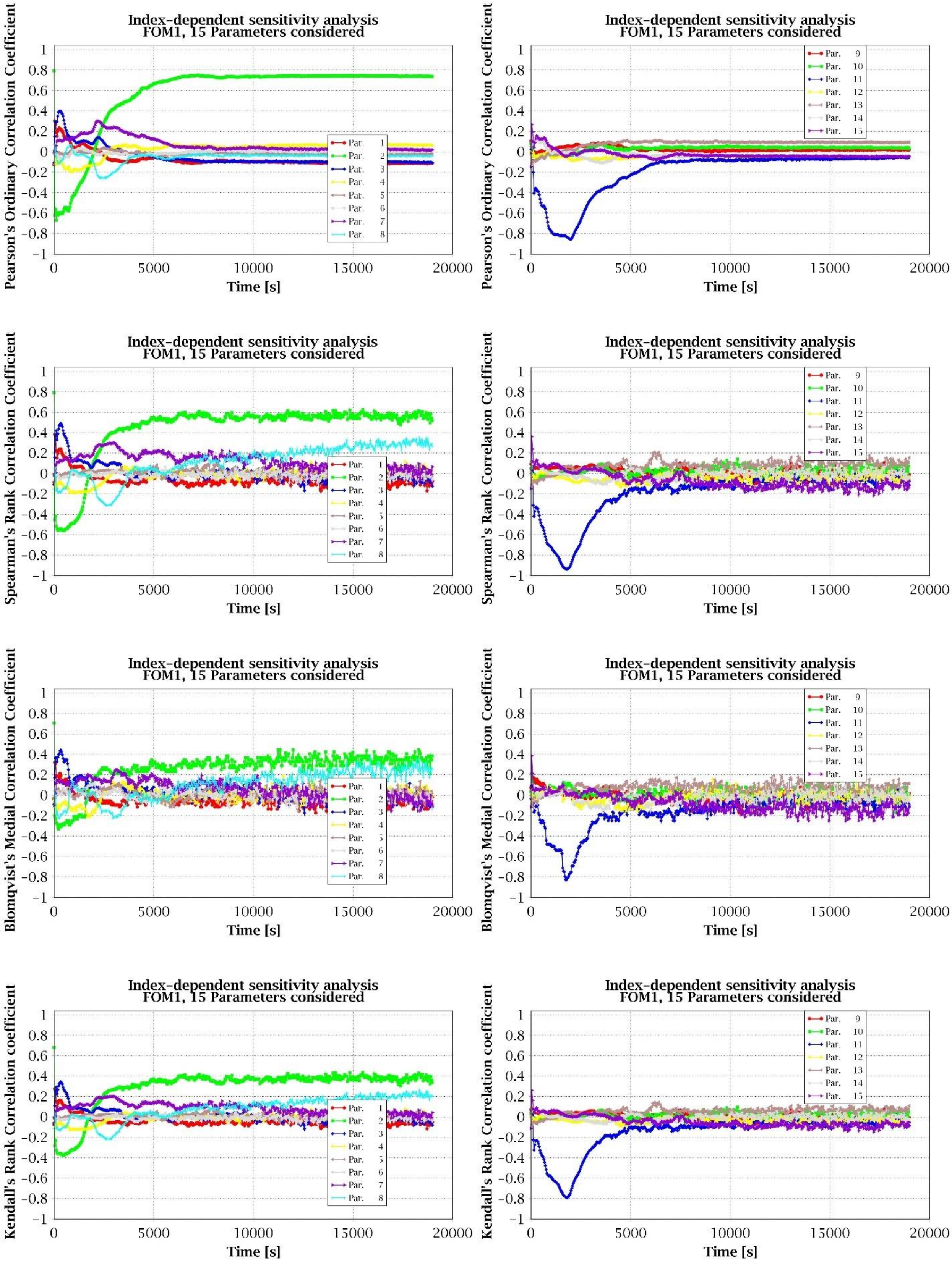

Figure 3-15 Primary Pressure – Correlation Based Sensitivity Indices

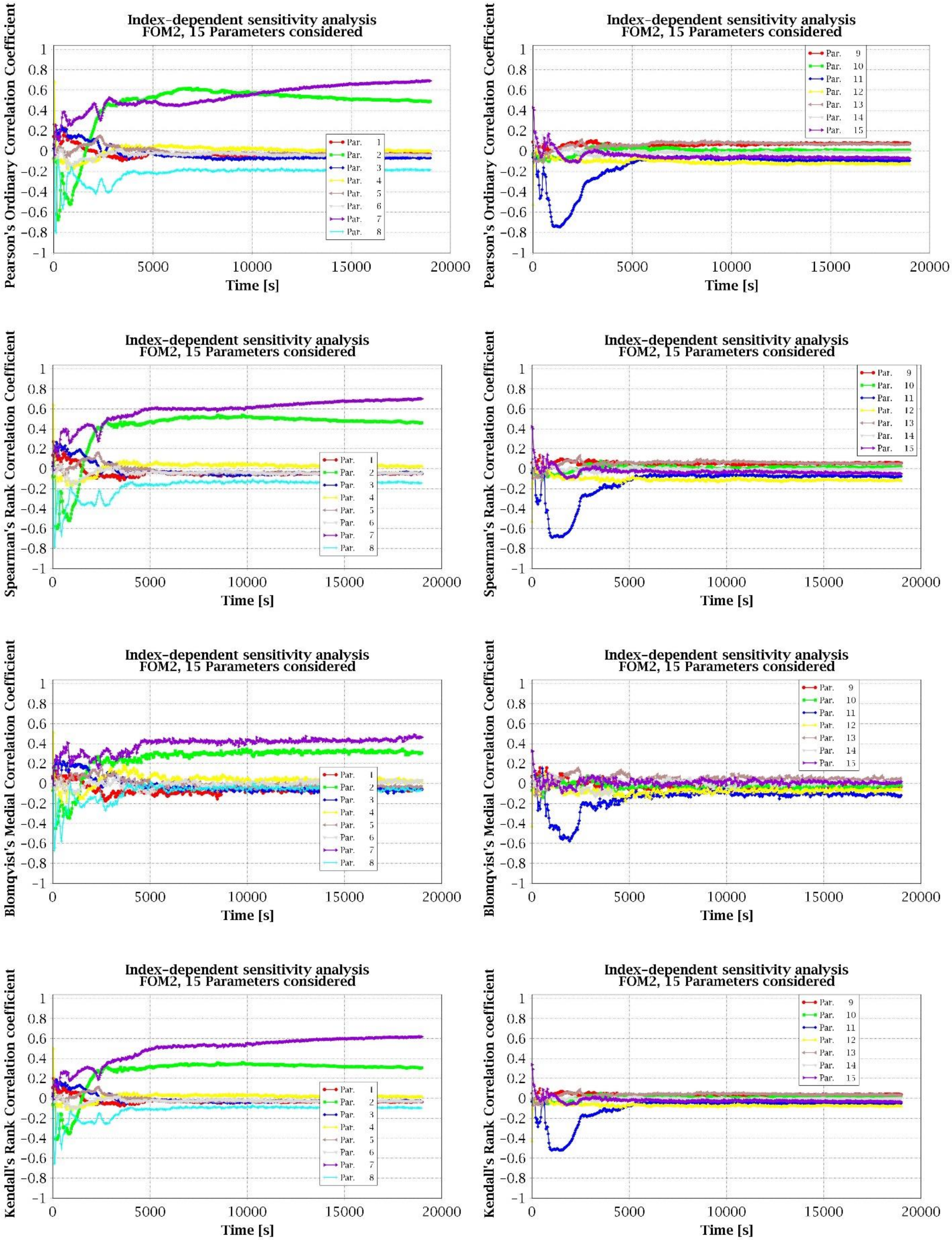

Figure 3-16 Liquid Temperature below RPV Cold Leg Inlet – Correlation Based Sensitivity Indices

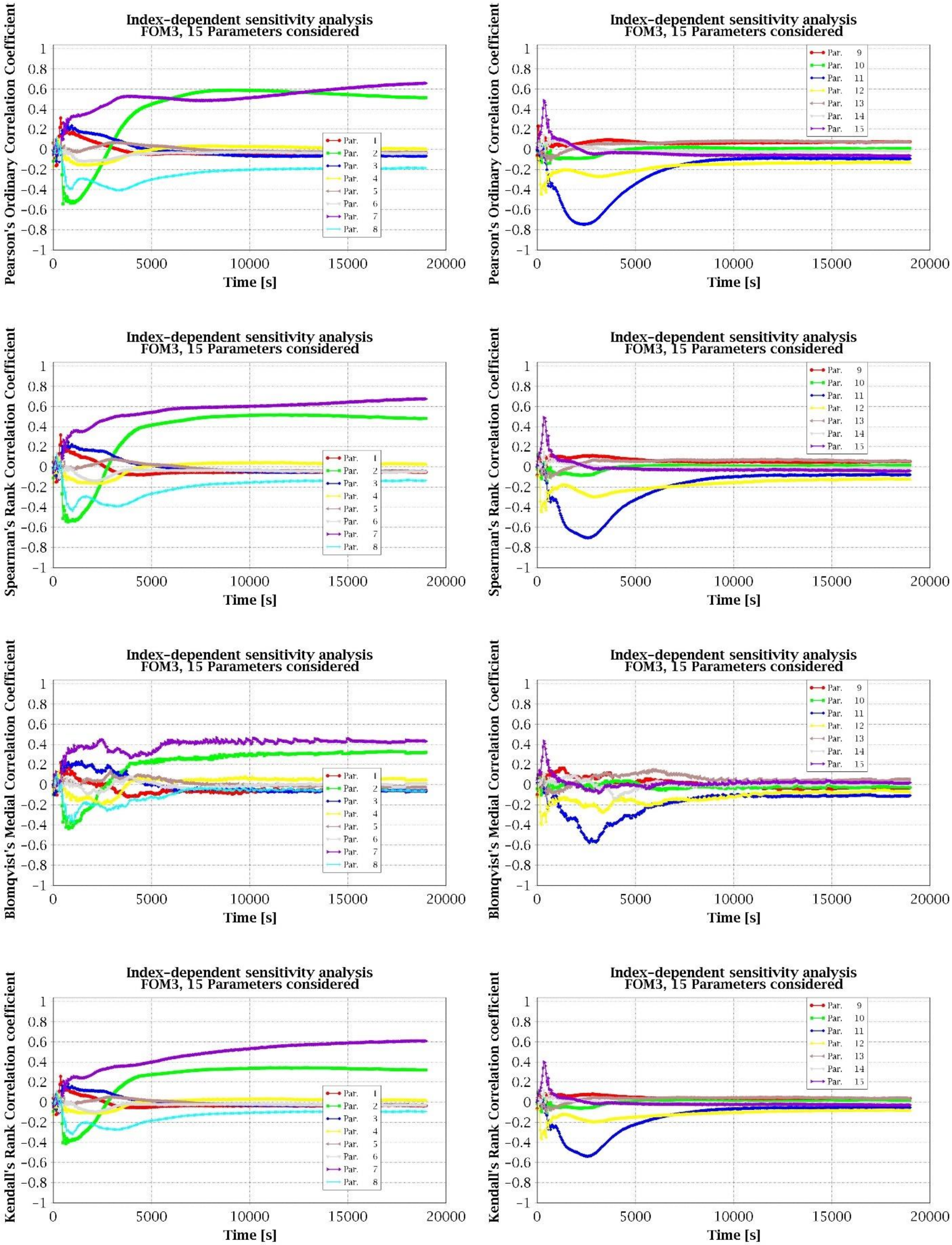

Figure 3-17 Wall Temperature below RPV Cold Leg Inlet – Correlation Based Sensitivity Indices

Sobol's indices show that strong influence on variance *Y* of FOM1 have parameters 'Par 2' and 'Par 11'. This agrees well with the results of the sensitivity study.

Sobol's indices show that strong influence on variance *Y* of FOM2 and FOM3 have parameters 'Par 2', 'Par 7' and 'Par 11'. This also agrees well with the results of sensitivity study.

The results of sensitivity study and sensitivity analysis qualitatively agree between each other.

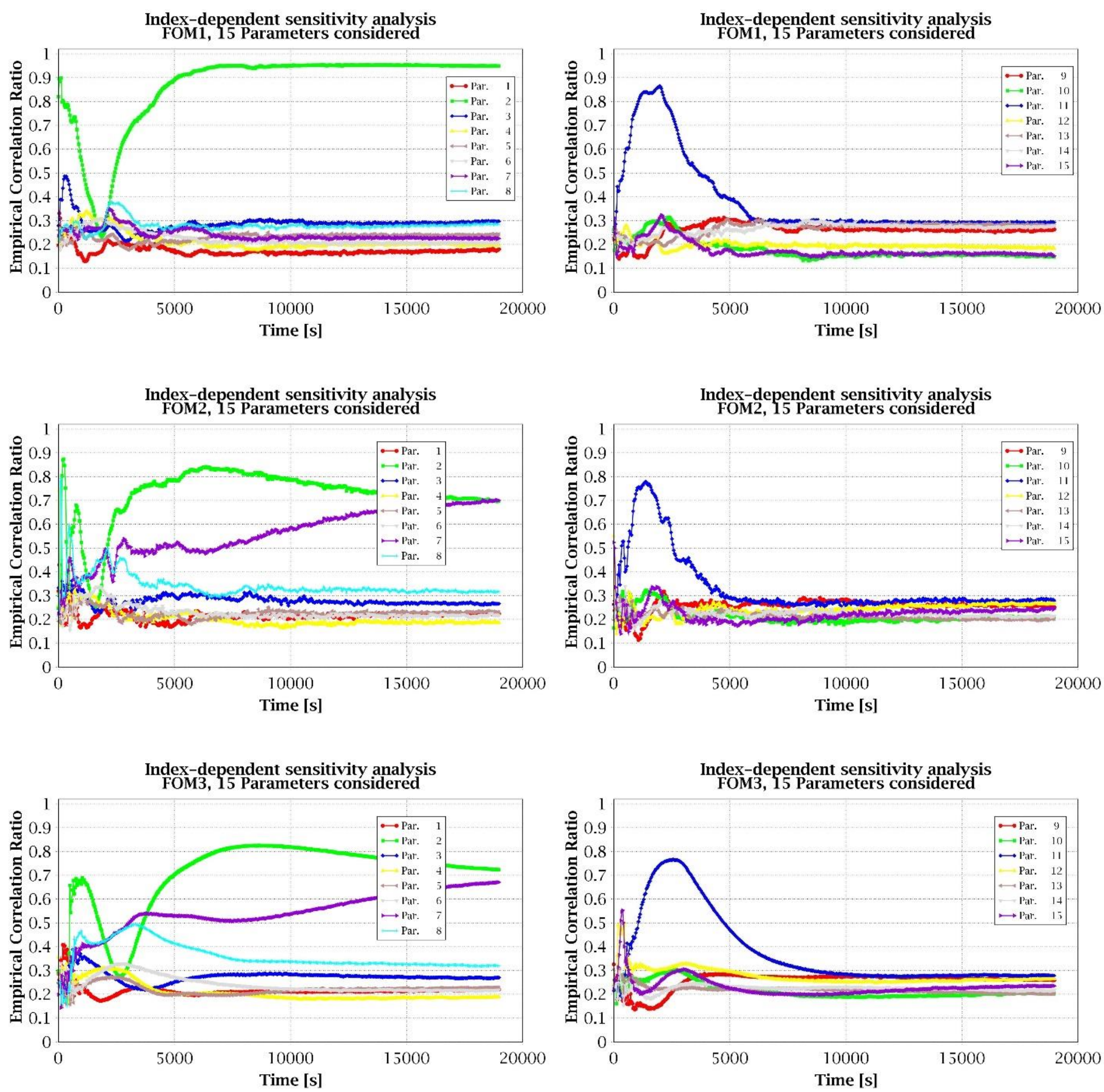

Figure 3-18 Sobol's Indices for the Selected FOMs

4 CONCLUSIONS

Uncertainty and sensitivity analysis of hot leg loss of coolant accident in two-loop pressurized water reactor, Westinghouse type, has been performed for RELAP5 version 3.3lj calculations. The selection of input uncertain parameters and distributions has been based on the results of the APAL project. In total one reference run and 208 sampled runs have been performed. In the uncertainty analysis minimum and maximum values in each time step were determined, representing lower and upper uncertainty bound, respectively. In the sensitivity analysis the four correlation based sensitivity indices and Sobol's first order index for output uncertain parameters were determined.

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NRC FORM 335 (12-2010) NRCMD 3.7 **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) **NUREG/IA-0547**	
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10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

The reactor pressure vessel (RPV) is vulnerable to occurrence of pressurised thermal shock (PTS) in a pressurised water reactor (PWR). PTS can occur during several overcooling scenarios, including loss of coolant accidents. Thermal hydraulic calculations of overcooling scenarios provide an input to structural analysis. The purpose of this study is to perform the calculation of hot leg small break loss of coolant accident with uncertainty and sensitivity analysis. The reactor selected was two-loop PWR, for which verified and validated input deck was available. For analysis the RELAP5 developmental version 3.3lj from 2022 with built-in code uncertainty parameters has been used. For uncertainty and sensitivity analysis Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) Data Processing program Software for Uncertainty and Sensitivity Analyses (SUSA) Version 4.2.5 has been used. For uncertainty and sensitivity analysis 208 RELAP5 runs have been performed. The three figures of merit selected for uncertainty analysis were reactor pressure, liquid temperature and reactor vessel wall temperature below the cold leg connection.

The results are presented for reference calculation, sensitivity study varying one parameter at a time and uncertainty and sensitivity analysis.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT unlimited
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April 2024